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CAROLINA POWER & LIGHT COMPANY
BRUNSWICK STEAM ELECTRIC PLANT

UNIT 0

EMERGENCY RADIATION WORK PERMITS

PLANT EMERGENCY PROCEDURE: PEP-03.3.5

VOLUME XIII

Rev. 000

Recommended By: G. E. Boyer Date: 4/16/84
Director - Administrative Support

Approved By: C. Dietz Date: 4/18/84
General Manager

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LIST OF EFFECTIVE PAGES

PEP-03.6.5

Page(s)	Revision
1-5	0

1.0 Responsible Individuals and Objectives

The Plant Monitoring Team is responsible for issuing required Radiation Work Permits in declared emergencies.

Individual workers and team leaders are responsible to the Site Emergency Coordinator for ensuring that emergency worker exposures are maintained within the guidelines of this procedure and ALARA to the extent possible.

2.0 Scope and Applicability

This procedure shall be implemented following declaration of an alert, site, or general emergency. Exhibit 3.3.5-1, Guidelines for Control of Personnel Radiation Exposure, provides instructions.

3.0 Actions and Limitations

3.1 Actions of all personnel prior to entering a high radiation area during emergency situations

3.1.1 Obtain and complete a Radiation Work Permit (RWP) in accordance with E&RC-0230, Issue and Use of Radiation Work Permit.

3.1.2 Obtain all special equipment and dosimetry specified on the RWP from the Personnel Protection and Decontamination Team.

3.1.3 Read and follow all instructions on the RWP.

3.2 Actions of the Plant Monitoring Team

3.2.1 Prepare the RWP in accordance with E&RC-0230, Issue and Use of Radiation Work Permit.

NOTE: At least a typical set of anti-Cs (see Exhibit 3.7.3-1) will be required for personnel performing the following actions/missions during an emergency:

- Sampling Reactor Coolant System fluids
- Sampling radioactive wastes (liquids, gases, etc.)
- Clean up of radioactive spills or contamination
- Entering an area of greater than 10 MPC airborne contamination
- Entering a radiation area of unknown intensity or contamination
- Entering the drywell
- Initial entries

Deviation from a full set of anti-Cs shall be approved by the Radiological Control Director.

3.2.2 As directed by the Plant Monitoring Team Leader, specify a high range dosimeter when:

3.2.2.1 Entering a radiation field of ≥ 10 R/hr.

3.2.2.2 Entering a radiation field of unknown intensity.

3.2.3 As directed by the Plant Monitoring Team Leader, specify finger badges when:

3.2.3.1 Handling radioactive material where expected extremity dose rate is ≥ 100 R/hr.

3.2.3.2 Working on pipes or equipment where expected extremity dose rate is ≥ 25 R/hr.

3.2.4 Record <u>any</u> <u>and</u> <u>all</u> additional dosimetry on the RWP for each person entering the radiation area. Whenever possible, finger badges should be labelled with the individual's security badge number and dosimeter serial numbers should be recorded on the RWP. Having this information available will facilitate data input into the RIMS computer.

3.2.5 Obtain authorization for the RWP from the Site Emergency Coordinator, General Manager, or the Manager - E&RC when exposures are expected to exceed the limits set forth in the 10CFR20 (> 3 rem/quarter).

3.2.6 The Site Emergency Coordinator may, at his discretion and as conditions warrant, defer requirements for a RWP, or portions thereof, prior to entry into a radiation area and give his authorization verbally.

3.2.6.1 A RWP shall be completed or a RIMS computerized RWP shall be completed by the individuals making a verbally authorized entry, as time permits, after the entry.

<u>NOTE</u>: Any person that has received a whole body dose totalling ≥ 5 rem by TLD for the year shall not be permitted to enter a controlled radiation area without the approval of the Site Emergency Coordinator or Manager - E&RC.

EXHIBIT 3.3.5-1

GUIDELINES FOR CONTROL OF PERSONNEL RADIATION EXPOSURE

Although an emergency situation transcends the normal requirements for limiting exposures to ionizing radiation, guideline levels are established for exposures that may be acceptable in emergencies. The maximum whole body dose received by any worker should not exceed established regulatory limits. Every reasonable effort will be used to ensure that an emergency is handled in such a manner that no worker exceeds these limits, including the administering of radioprotective drugs where recommended by expert medical opinion. The acceptability of higher exposures is restricted to emergency situations where some clear and definite advantage can be gained by such worker exposure. It is compatible with the risk concept to accept exposures leading to doses considerably in excess of those appropriate for normal occupational difficulty, if necessary. The saving of life, measures to circumvent substantial exposures to population groups, or the preservation of valuable installations may all be sufficient cause for accepting above normal exposures. These higher dose limits cannot be specified; however, they should be commensurate with the significance of the objective and held to the lowest practicable level. As discussed below, all planned exposures should follow the guidelines set forth in Report 39 of the National Council on Radiation Protection, specifically paragraphs 257 through 259 of that report, which deal with planned occupational exposure under emergency conditions.

Decision making is based on conditions at the time of an emergency and should always consider the probable effects of an exposure prior to allowing any individual to be exposed to radiation levels exceeding the established occupational limits. The probable high radiation exposure effects are:

1. Up to 50 rem in one day - No physiological changes are likely to be observed.

2. 50-100 rem - No impairment likely but some physiological changes, including possible temporary blood changes, may occur. Medical observations will be required after exposure.

3. 100-300 rem - Some physical impairment possible; some lethal exposures possible.

The following subsections describe the criteria to be considered for lifesaving and facility protection actions.

Lifesaving Actions*

In emergency situations that require personnel to search for and remove injured persons or entry to prevent conditions that would probably injure numbers of people, a planned dose shall not exceed 100 rem to the whole body and a planned additional dose of up to 200 rem (i.e., a total of 300 rem) to the hands, forearms, feet, and ankles. The following additional criteria should be considered:

1. Rescue personnel should be volunteers or professional rescue personnel (i.e., fire fighters or first aid and rescue personnel who volunteer by choice of employment).

2. Rescue personnel should be broadly familiar with the probable consequences of exposure.

3. Women capable of reproduction should not take part in these actions.

4. Other things being equal, volunteers above the age of 45 should be selected whenever possible for the purpose of avoiding unnecessary genetic effects.

5. Internal exposure should be minimized by the use of the most appropriate respiratory protection and contamination should be controlled by the use of protective clothing when practical.

6. Exposure under these conditions shall be limited to once in a lifetime.

7. Persons receiving exposures as indicated above should avoid procreation for a period up to a few months.

Exposure During Reentry/Repair Efforts

There may be situations where saving a life is not at issue but where it is necessary to enter a hazardous area to protect valuable installations or to make the facility more secure against events which could lead to radioactive releases (i.e., assessment actions or entry of damage repair parties who are to repair valve leaks or add iodine-fixing chemicals to spilled liquids). In such instances, planned dose to emergency workers should not exceed 25 rem to the whole body, 125 rem to the thyroid, or 100 rem to the extremities. The following additional criteria should also be considered:

1. Persons performing the planned actions should be volunteers broadly familiar with exposure consequences.

2. Women capable of reproduction will n t take part in these actions.

*This guideline applies to the removal of injured persons if the saving of life is possible or entry to prevent conditions that, if left uncorrected, could lead to damage or releases that would probably injure numbers of people on and off site.

3. Internal exposures shall be minimized by respiratory protection; contamination controlled by the use of protective clothing.

4. If the retrospective dose from these actions is a substantial fraction of the prospective limits, the actions shall be limited to once in a lifetime.

5. Entry into high radiation areas shall not be permitted unless instrumentation capable of reading radiation levels of up to 1000 R/hour (gamma) is provided.

6. Each emergency worker entering a high radiation area shall wear pocket dosimeters capable of measuring the expected exposure to be received.

7. Entry into radiation fields of greater than 100 R/hour shall not be permitted unless specifically authorized by the plant General Manager or Manager - E&RC; in their absence the Site Emergency Coordinator may grant approval.

8. Planned exposures in excess of 3 rem may only be approved by:

    a. Plant General Manager, or;

    b. Manager - E&RC, or

    c. Site Emergency Coordinator in their absence.

Emergency teams that must enter areas where they might be expected to receive higher than normal doses will be fully briefed regarding their duties and actions and what they are to do while in the area. They will also be fully briefed as to the expected dose rates, stay time, and other hazards. All such entries will include one member from the plant Monitoring Team or other person adequately trained in Health Physics. All team members will use clothing, dosimeters, respiratory devices, and other protective devices as specified by the Radiological Control Director. Team members will be instructed not to deviate from the planned route unless required by unanticipated conditions, such as rescue or performing an operation that would minimize the emergency condition. If monitored dose rates or stay times encountered during the entry exceed the limits set forth for the operation, the team will immediately communicate with the Site Emergency Coordinator, the Radiological Control Director, or will return to the area from where they were dispatched.

Once their operation has been completed, team personnel will follow established monitoring and personnel decontamination procedures or as specified by the Radiological Control Director.

CAROLINA POWER & LIGHT COMPANY
BRUNSWICK STEAM ELECTRIC PLANT

UNIT 0

CONFIRMATION OF OFF-SITE DOSE PROJECTIONS

PLANT EMERGENCY PROCEDURE: PEP-03.5.1

VOLUME XIII

Rev. 004

Recommended By: G E Boyer Date: 4/16/84
Director - Administrative Support

Approved By: C Dietz Date: 4/18/84
General Manager

LIST OF EFFECTIVE PAGES

PEP-03.5.1

PAGE(S)	REVISION
1-4	4

1.0 Responsible Individual and Objectives

The initial calculations of the consequences of an accidental release are necessarily based on estimated release rates and atmospheric dispersion. Uncertainties in these estimates can result in calculations which differ by an order of magnitude from the actual off-site consequences. Confirmation and/or modification of the dose projections may be required before a decision is made to notify the public or initiate off-site protective actions.

The Environmental Monitoring Team is responsible to and shall report to the Radiological Control Director until the Emergency Operations Facility is activated and fully staffed. At this time, the Radiological Control Manager in the Emergency Operations Facility shall assume all responsibility for the Environmental Monitoring Teams, the interpretation of off-site data, and its comparisons with dose projections.

When Corporate Monitoring Teams arrive and are able to assume the environmental monitoring functions, the Plant Environmental Monitoring Team may then return to the plant under the direction of the Radiological Control Director or remain in active support of EOF personnel if so directed by the Radiological Control Director.

2.0 Scope and Applicability

This procedure provides guidelines for the location of environmental measurements, the measurements to be taken and the comparison of measured radiation levels with the dose projections developed by the Dose Projection Team.

This procedure should be implemented immediately upon declaration of any emergency class where a release of radioactivity to the atmosphere has occurred or is believed to have occurred. It may be used to confirm that meteorological dispersion estimates are valid, where more detailed consequences have been developed.

3.0 Actions and Limitations

3.1 The Environmental Monitoring Team shall:

3.1.1 Consult with the Environmental Monitoring Team Leader and obtain the current wind direction data or areas to be surveyed. Upon activation, this information should be obtained from the Radiological Control Manager.

Note: Wind direction data is normally reported as direction from which the wind is blowing, so that off-site surveys are in the opposite direction and downwind. Confirm wind direction.

3.1.2 If the area to be surveyed is not specified, the following guidelines apply:

WIND DIRECTION BETWEEN:	SURVEY LOCATION
270° and 0° (between W and N)	NCSR 1527 to River Road to NCSR 1528
0° and 45° (between N and NE)	NCSR 1526 to Old River Road to NCSR 1527
45° and 90° (between NE and E)	NC87 to NCSR 1526
90° and 135° (between E and SE)	NC87 to NC133 (towards Wilmington)
135° and 180° (between SE and S)	NC133 to NCSR 1525
180° and 225° (between S and SW)	NCSR 1525
225° and 270° (between SW and W)	Primary: Land vehicle at about 1300 meters. Secondary: Cape Fear River from Snow Marsh north along Sunny Point Army Terminal

These are shown on the Operations map.

3.1.3 If weather conditions do not permit monitoring at ground level or on the river, advise Radiological Control Director (Radiological Control Manager after Emergency Operations Facility has been activated) that helicopter assistance may be needed.

3.1.4 Once the initial survey location is identified, pick up survey gear from environmental kits located at the Visitors Center. (E&RC-0600, Appendix G lists the contents of the kit.)

3.1.5 Request, from the Radiological Control Director, (Radiological Control Manager after Emergency Operations Facility has been activated) information on expected radiation conditions to be encountered and on any special protective gear required.

3.1.6 Proceed to the survey vehicle, load the survey equipment and establish communications with the Environmental Monitoring Team Leader.

3.1.7 Proceed to the survey location.

3.1.8 Perform dose rate surveys to assess noble gas release.

3.1.9 Perform survey in accordance with guidelines outlined in E&RC-3215, Field Estimate of Airborne I-131 Concentration, or E&RC-3217, Field Estimate of Airborne Particulate Concentration.

3.1.10 Proceed as directed, to the site to return the samples for analysis or to other survey locations.

3.2 The Environmental Monitoring Team Leader (in consultation with the Dose Projection Coordinator) shall:

3.2.1 Compare the maximum off-site dose rate readings to the projected whole body doses based on plant measurements of noble gas releases and estimated meteorological conditions.

NOTE: The actual meteorological dispersion values, for any given meteorological stability class, may vary by a factor of five or even more as compared with the values based on standard tables or figures. Where the observed dose rates are within a factor of five of the calculated dose rates it may be assumed that the initial dose projections are reasonably representative of the consequences of the release.

3.2.2 If survey meters held against the samples indicate activity has been retained on the filters, this may be evidence that iodine has been released. If the activity on both filters at the second reported reading are within 25% of the first reading it should be presumed, pending isotopic analysis, that iodine is present.

NOTE:

1. Noble gases will be retained to some extent on charcoal cartridges. It will slowly off-gas. Rb-88, with a 17 minute half-life, may be the predominant activity on paper filters. Thus activity on air samples may be the result of a noble gas release. The above step is an attempt to quickly determine whether iodine has also been released.

2. Each of the emergency sirens throughout the counties has an electrical outlet which can be used to run air samplers. Each of these sirens is numbered. (See Appendix B for maps with siren locations.)

3.3 If requested, the Environmental Monitoring Team shall brief state monitoring teams regarding conditions found prior to their activation.

NOTE: It is very important to identify whether there was confirmation of the presence or of the absence of radio iodine in the environment.

CAROLINA POWER & LIGHT COMPANY
BRUNSWICK STEAM ELECTRIC PLANT

UNIT 0

EXPANDED ENVIRONMENTAL MONITORING

PLANT EMERGENCY PROCEDURE: PEP-03.5.2

VOLUME XIII

Rev. 002

Recommended By: L C Boyer Director - Administrative Support Date: 4/16/84

Approved By: C Dietz General Manager Date: 4/18/84

LIST OF EFFECTIVE PAGES

PEP-03.5.2

Page(s)	Revision
1-2	2

1.0 Responsible Individual and Objectives

The Environmental Monitoring Team is responsible to the Radiological Control Director (the Radiological Control Manager after the Emergency Operations Facility is activated) for conducting environmental surveys and the placement and collection of environmental samplers in the event of an accidental release of radioactive material from the plant. The surveying and sampling will be greater in extent and frequency than during routine operations.

2.0 Scope and Applicability

This procedure includes all CP&L environmental monitoring at and beyond the protected area fence. This procedure should be implemented in parallel with PEP-03.5.1, "Confirmation of Off-site Dose Projections." Where manpower resources are limited, implementation of this procedure may be deferred until PEP-03.5.1 has been completed. This procedure is not intended to replace any state- or county-directed efforts to determine levels of radioactivity in the environment, although it may provide the basis for initial assessments by public agencies.

NOTE: This procedure should be performed in conjunction with E&RC-3110.

3.0 Actions and Limitations

3.1 The Environmental Monitoring Team shall, as directed:

3.1.1 Place additional TLDs approximately every 10 meters around the exclusion area perimeter in the sector within 45° of the plume centerline.

3.1.2 Place TLDs along the road surrounding the site in the sector within plus or minus 22.5° of the plume centerline (a total of sampling arc of 45°).

NOTE: The spacing of these TLDs should be placed about 50 meters apart to permit improved assessment of the concentrations of radioactivity in the environment and provide an important baseline for verifying source term estimates.

3.1.3 Remove, replace, and supplement these as directed by the Radiological Control Director (the Radiological Control Manager after the Emergency Operations Facility is activated).

3.1.4 As soon as practicable, and thereafter as directed by the Radiological Control Director (the Radiological Control Manager after the Emergency Operations Facility is activated), remove and change all routine air particulate and charcoal filters and all routine TLDs. Location of these samples are included in E&RC-3110.

CAUTION: IN COLLECTING ANY ENVIRONMENTAL SAMPLES, TAKE CARE TO PREVENT CROSS-CONTAMINATION OF SAMPLES.

3.1.5 Where releases of materials other than noble gas are known or are believed to have occurred, collection of vegetation, milk, or other substances may be appropriate. Any sampling of such media shall be coordinated with and be under the general direction of responsible state officials at the State Emergency Response Team Headquarters.

3.2 In the event of liquid releases to the discharge canal, collect samples as per routine environmental sampling procedures but with frequencies as directed by the Radiological Control Director (the Radiological Control Manager after the Emergency Operations Facility is activated).

3.3 Unless otherwise specified by the Radiological Control Director (the Radiological Control Manager after the Emergency Operations Facility is activated), samples should be collected in accordance with existing usual procedures.

CAROLINA POWER & LIGHT COMPANY
BRUNSWICK STEAM ELECTRIC PLANT

UNIT 0

PLUME TRACKING BY ACTUAL MEASUREMENT

PLANT EMERGENCY PROCEDURE: PEP-03.5.3

VOLUME XIII

Rev 003

Recommended By: L. C. Boyer
Director - Administrative Support
Date: 4/16/84

Approved By: C. Dietz
General Manager
Date: 4/18/84

LIST OF EFFECTIVE PAGES

PEP-03.5.3

Page(s)	Revision
1 - 3	3

PEP-3.5.3 PLUME TRACKING BY ACTUAL MEASUREMENT

1.0 Responsible Individual and Objectives

The Radiological Control Director (the Radiological Control Manager after the Emergency Operations Facility is activated) is responsible for maintaining up-to-date assessment of the areas affected by radioactivity release into the environment.

2.0 Scope and Applicability

This procedure is to be implemented when projected off-site exposures approach or exceed the levels associated with Environmental Protection Agency's Protective Action Guidelines. A determination must be made of doses in the affected areas. While these doses may be inferred from interpretations of release data, time varying meteorological conditions and results from environmental surveys, actual plume tracking should be attempted where practicable. This procedure applies primarily to CP&L activities, necessary for emergency responses, but performed prior to full activation of the state response organizations. Where releases continue for more than several hours, plume tracking efforts will be under the direction of the State, but CP&L may be requested to provide input. This procedure also addresses CP&L support of state-directed plume tracking efforts.

3.0 Actions and Limitations

3.1 The Environmental Monitoring Team shall provide necessary personnel and equipment to measure radioactivity levels in the plume.

3.1.1 Obtain maps, portable survey equipment, including an air sampler and a survey meter.

3.1.2 Consult with the Environmental Monitoring Team Leader (Radiological Control Manager after the Emergency Operations Facility is activated) and review the release, estimated release heights and wind directions.

3.1.3 Proceed to the plume tracking vehicle and establish communications with the Environmental Monitoring Team Leader.

3.1.4 Proceed to a distance approximately 1 km (0.6 miles) from the plant, in the general downwind direction.

3.1.5 Travel at a right angle to the reported wind direction and measure the highest dose rate.

3.1.6 If it has been determined that the release is from an elevated location or is associated with steam (as from open steam relief valve, steam line break, etc.) attempt (where it is safe to do so) to measure the height at which the maximum dose rate is observed.

NOTE: This will provide useful benchmark information to help interpret subsequent measurements.

3.1.7 Proceed downwind, periodically taking crosswind measurements and reporting locations of maximum readings at any given distance.

NOTE: This may need to be repeated for continuing releases.

3.1.8 Collect air samples in accordance with E&RC-3215, Field Estimate of Airborne I-131 Concentration or E&RC-3217, Field Estimate of Airborne Particulate Concentration.

3.2 The Environmental Monitoring Team Leader shall:

3.2.1 Consult the Dose Projection Coordinator to obtain projected plume trajectory estimates.

3.2.2 Record and display a summary of the reported results (time, location, rate) on the map.

3.2.3 If releases or meteorological conditions change substantially during the plume tracking effort, advise and make recommendations as to revised tracking locations.

3.2.4 If plume tracking results indicate doses significantly different than those projected, suggest that the dose projections be revised.

NOTE: Make sure that all reported dose rates are adjusted to reflect a common time (e.g., correct for differences in decay) before revising projected doses.

3.2.5 If deemed necessary and if not already done by the State, request assistance from the Department of Energy for special plume surveys.

NOTE: This cannot be made available for a number of hours, but can be extremely useful in measuring low levels of radioactivity. Such requests should be made through the North Carolina State Radiation Protection Section.

3.2.6 If assessments of the plume trajectory are uncertain and if not already done so by the State, request assistance from NOAA and/or the NWS.

NOTE: These groups may be able to provide expert advice on regional meteorological conditions. Depending on the existing conditions, they may be able to provide devices to measure existing wind patterns (sounding balloons, theodolites) and thus plume trajectories. Such requests should be made through the North Carolina Radiation Protection Section.

CAROLINA POWER & LIGHT COMPANY
BRUNSWICK STEAM ELECTRIC PLANT

UNIT 0

ESTIMATE OF THE EXTENT OF CORE DAMAGE UNDER ACCIDENT CONDITIONS

PLANT EMERGENCY PROCEDURE PEP-03.6.3

VOLUME XIII

Rev. 003

Reviewed By: William R. Zoller Date: 4-9-84
QA

Recommended By: L. C. Boyer Date: 4/16/84
Director - Administrative Support

Approved By: C. Dietz Date: 4/18/84
General Manager

LIST OF EFFECTIVE PAGES

PEP-03.6.3

Page(s)	Revision
1-34	3

1.0 Responsible Individual and Objectives

The Radiological Control Director is responsible to the Site Emergency Coordinator for determining the magnitude and rate of potential radioactive releases to the environment. The Radiological Control Director may delegate the calculational aspects to the Plant Sampling and Analysis Team Leader.

The Accident Assessment Team Leader should be familiar with this procedure and available for consultation as requested by the Dose Projection Coordinator.

2.0 Scope and Applicability

This procedure is to be implemented by the Site Emergency Coordinator or the Radiological Control Director whenever the potential for core damage exists and/or there exists a potential or actual radiological release to the environment (e.g., site or general emergency).

This procedure provides information on inventories of reactor full-power radioisotopes in curies and gives methods for comparing actual radioactive liquid and gaseous samples with expected activity levels after a reactor accident based on cesium, noble gases, and iodines. There are several other plant parameters which are measured in the BWR which can provide sufficient information to confirm the initial core damage estimate based on radionuclide measurements.

Containment radiation level provides a measure of core damage, because it is an indication of the inventory of airborne fission products (i.e., noble gases, a fraction of the halogens, and a much smaller fraction of the particulates) released from the fuel to the containment. Containment hydrogen levels, which are measurable by the PASS or the containment gas analyzers, provide a measure of the extent of metal water reaction which, in turn, can be used to estimate the degree of clad damage.

Another significant parameter for the estimation of core damage is reactor vessel water level. This parameter is used to establish if there has been an interruption of adequate core cooling. Significant periods with the core uncovered, as evidenced by reactor vessel water level readings, would be an indicator of a situation where core damage is likely. Water level measurement would be particularly useful in distinguishing between bulk core damage situations caused by loss of adequate cooling to the entire core and localized core damage situations caused by a flow blockage in some portion of the core.

There are other parameters which may provide an indication that a core damage event has occurred. These are main steam line radiation level and reactor vessel pressure. The usefulness of main steam line radiation measurement is limited because the main steam line radiation monitors are

downstream of the main steam isolation valves (MSIVs) and would be unavailable following vessel isolation. Reactor vessel pressure measurement would provide an ambiguous indication of core damage, because, although a high reactor vessel pressure may be indicative of a core damage event, there are many nondegraded core events which could also result in high reactor vessel pressure.

There are other measurements besides radionuclide measurements which are obtainable using the PASS which would further aid in estimating core damage. Detection of such elements in the reactor coolant as Sr, Ba, La, and Ru is evidence of fuel melting. These indications could be factored into the final core damage estimate.

3.0 Actions and Limitations

3.1 Summary of Method

Liquid and gaseous samples will be obtained from the Postaccident Sampling System (PASS)--Liquid from the reactor coolant and/or suppression pool and gaseous samples from the primary and/or secondary containment. The samples will be quantitatively analyzed on the appropriate equipment. The results of the above analysis, in addition to containment radiation level, hydrogen analysis, and the core water level history, will be used in the estimation. This procedure follows the General Electric procedure NEDO-22215, August 1982.

List of Exhibits

3.2 Limitations

3.2.1 Analysis of PASS samples for concentrations of Ba, Sr, La, and Ru and consideration of the relative amounts of fission products would indicate if any fuel melt has occurred.

3.2.2 The selection of a sample location should account for the type of event which will determine where the fission products will concentrate.

3.2.3 The recommended sampling locations are as follows:

Gaseous

Event Type	Sample Location
Nonbreaks (e.g., MSIV)	Suppression pool atmosphere
Small breaks	Drywell (before depressurization); suppression pool atmosphere (after depressurization)
Large breaks (liquid or steam) in primary containment	Drywell
Large breaks outside primary containment	Suppression pool atmosphere

3.2.4 The recommended sampling location for liquid for all events is the jet pumps as long as there is sufficient reactor pressure (normally > 50 psig) to provide a sample from that location. If there is not sufficient reactor pressure to allow a sample to be taken from the jet pumps, the sample should be taken from the sample points on the RHR System.

3.2.5 If a jet pump liquid sample is requested at low (< 1%) power conditions for a small break or nonbreak event, recommend to Operations that the reactor water level be raised to the level of the moisture separators. This will fully flood the moisture separators and will provide a thermally induced recirculation flow path for mixing.

3.3 Actions

3.3.1 Evaluations of Liquid and Gaseous Samples

NOTE: The extent of core damage can be determined by comparing the measure concentrations of major fission products in either the gas or water samples, after appropriate normalization, with the reference plant data.

3.3.1.1 The plant Sampling and Analysis Team Leader should request samples from the PASS.

NOTE: Steps 3.3.1.2 through 3.3.1.7 can be accomplished using PASS, a computer program developed for use on the

Dose Projection Team's IBM Personal Computer. To use the program, the Plant Sampling and Analysis Team Leader should complete Exhibit 3.6.3-7, Computer Inputs for the PASS Program, and give the completed exhibit to the Dose Projection Coordinator who will run the program and return the results. Exhibit 3.6.3-8 provides example test cases which can be used to verify that the computer program PASS is working properly. Expected results for known computer inputs are given. These test cases should be used to demonstrate the validity of PASS each time the program is initially used.

3.3.1.2 Obtain the samples from the PASS and determine the concentration of the fission product i (C_{wi} in water or C_{gi} in gas as determined in Appendix A using data provided in Exhibit 3.6.3-3).

3.3.1.3 Correct the measured concentration for decay to the time of reactor shutdown. Ensure that the measured gaseous activity concentration has been corrected for temperature and pressure difference in the sample vial and the containment (torus) gas phase.

NOTE: This is normally included in the quantitative analysis results.

3.3.1.4 Calculate the fission product inventory correction factor F_{Ii} per Appendix B and record on Worksheet A2.

3.3.1.5 Calculate the C_{wi} and C_{gi} using the information obtained in Step 3.3.1.2 and the methods in Appendix A and record on Worksheet A1.

3.3.1.6 Using the correction factors, determined in Appendices A and B, calculate the normalized concentration, C_{wi}^{Ref} or C_{gi}^{Ref}, per Appendix C and record on Worksheet A3.

3.3.1.7 Use Exhibit 3.6.3-2 to estimate the extent of fuel or cladding damage using C_{wi}^{Ref} for Cs-137 and I-131 and C_{gi}^{Ref} for Xe-133 and Kr-85. Record data on Worksheet A4.

3.3.2 <u>Evaluation of Metal-Water Reaction and Inventory Release</u>

3.3.2.1 Use Appendix D to determine the percent metal-water reaction. Record data on Worksheet B1.

3.3.2.2 Use Appendix E to determine the fuel inventory release to the containment. Record data on Worksheet B2.

3.3.3 <u>Application of Other Significant Parameters to Core Damage Estimate</u>

Section 3.3.1 provides an estimate of core damage based on radionuclide measurements. Based on Step 3.3.1.7, an initial assessment of core damage is made. Based on a clarification provided by the NRC, that assessment would appear in a matrix as follows:

Degree of Degradation	Minor (< 10%)	Intermediate (10% - 50%)	Major (> 50%)
No fuel damage	←	1	→
Cladding failure	2	3	4
Fuel overheat	5	6	7
Fuel melt	8	9	10

As recommended by the NRC, there are four general classes of damage and three degrees of damage within each of the classes except for the "no fuel damage" class. Consequently, there are a total of 10 possible damage assessment categories. For example, Category 3 would be descriptive of the condition where between 10% and 50% of the fuel cladding has failed. Note that the conditions of more than one category could exist simultaneously. The objective of the final core damage assessment procedure is to narrow down, to the maximum extent possible, those categories which apply to the actual in-plant situation.

The initial core damage assessment based on radionuclide measurement will provide one or several candidate categories which most likely represent the actual in-plant condition. The other parameters should then be evaluated (as identified in Section 3.3) to corroborate and further refine the initial estimate.

For example, fission product measurement using PASS may indicate Category 4 core damage and, additionally, the potential for fuel overheat and fuel melt (i.e., Categories 5 through 10). Measurement of hydrogen in containment and use of the hydrogen correlation provided in Appendix D is used to verify that extensive clad damage had occurred. Use of the containment radiation monitor reading along with the correlation provided in Appendix E would verify that a significant fission product release to the containment had occurred, further verifying the initial assessment.

Further analysis of the PASS samples for concentrations of Ba, Sr, La, and Ru and consideration of the relative amounts of fission products released would indicate if any fuel melt had occurred.

Exhibit 3.6.3-1 indicates how the analysis of the other significant parameters relates to the estimation of core damage based on radionuclide measurements.

3.3.4 Consult with the Dose Projection Coordinator and the Radiological Control Director when results of this procedure are determined and repeat this procedure as necessary.

4.0 References

Lin, C. C., "Procedure for the Determination of the Extent of Core Damage Under Accident Conditions," NEDO-22215, 1982.

Letter and Attachment from Mr. D. K. Smith, Service Supervisor - Nuclear, General Electric to Mr. A. C. Tollison, Jr., General Manager, Brunswick Steam & Electric Plant, dated November 9, 1979, Subject: Radiation Source Term Information.

EXHIBIT 3.6.3-1

SEQUENCE OF ANALYSIS FOR
ESTIMATION OF CORE DAMAGE

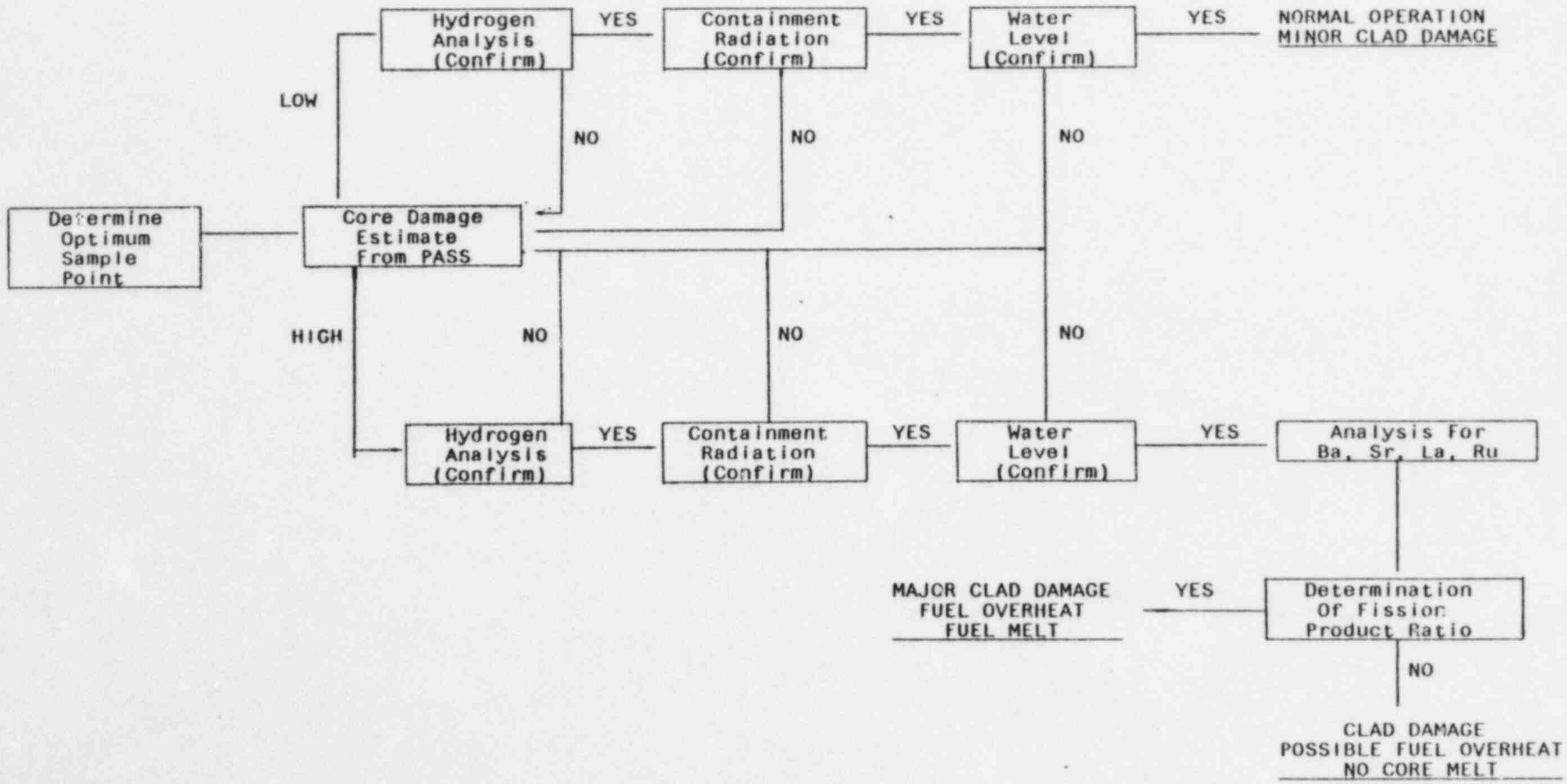
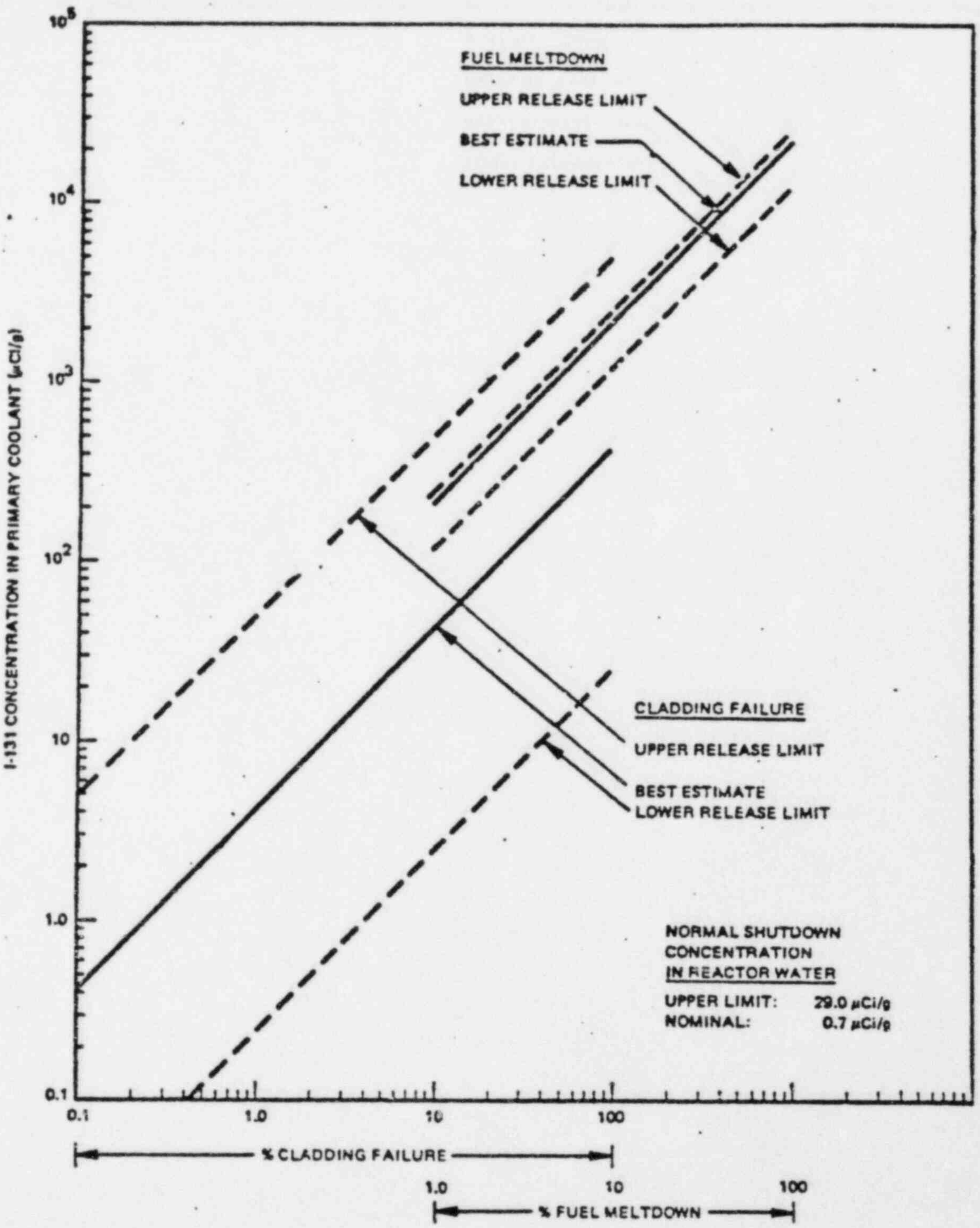
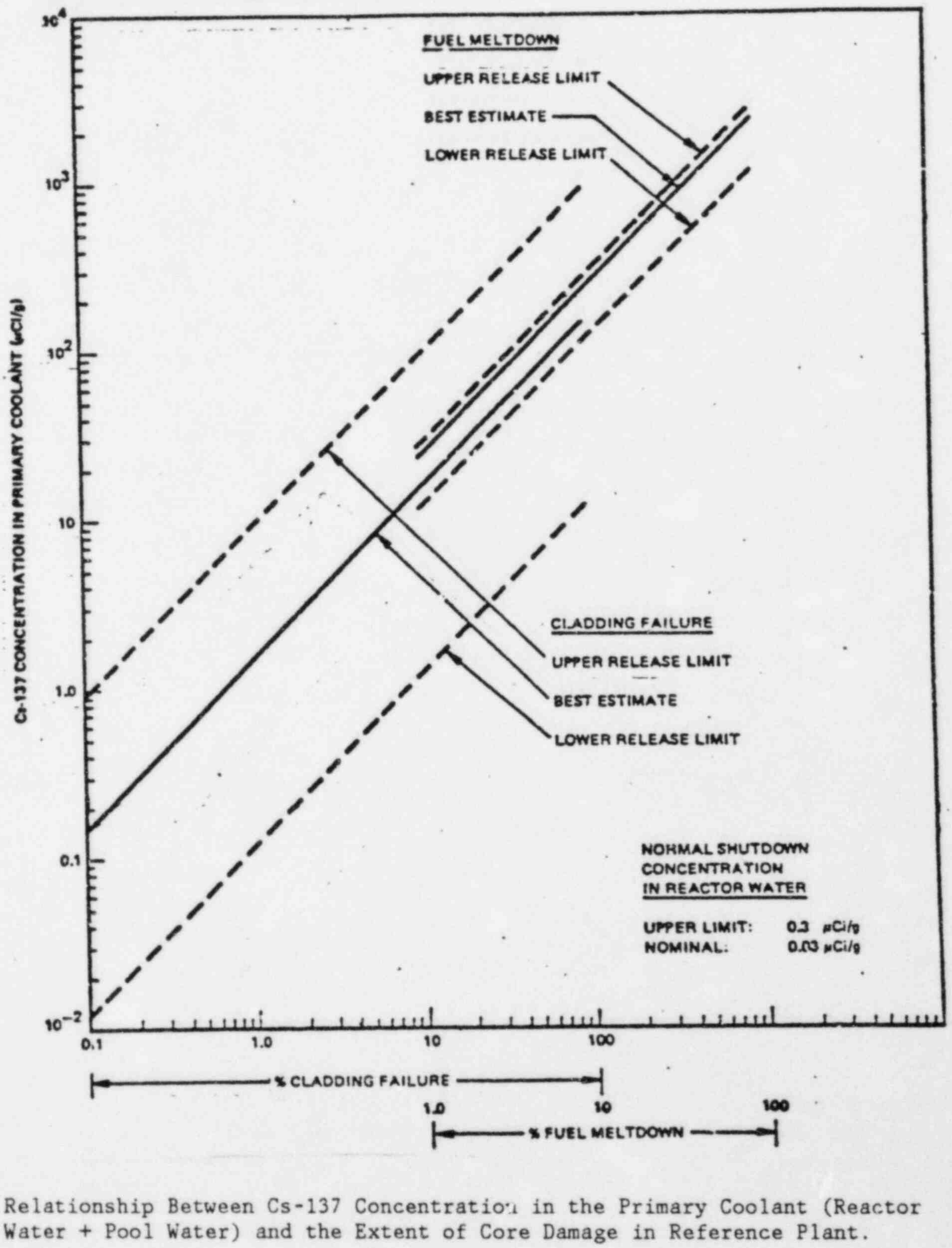


EXHIBIT 3.6.3-2



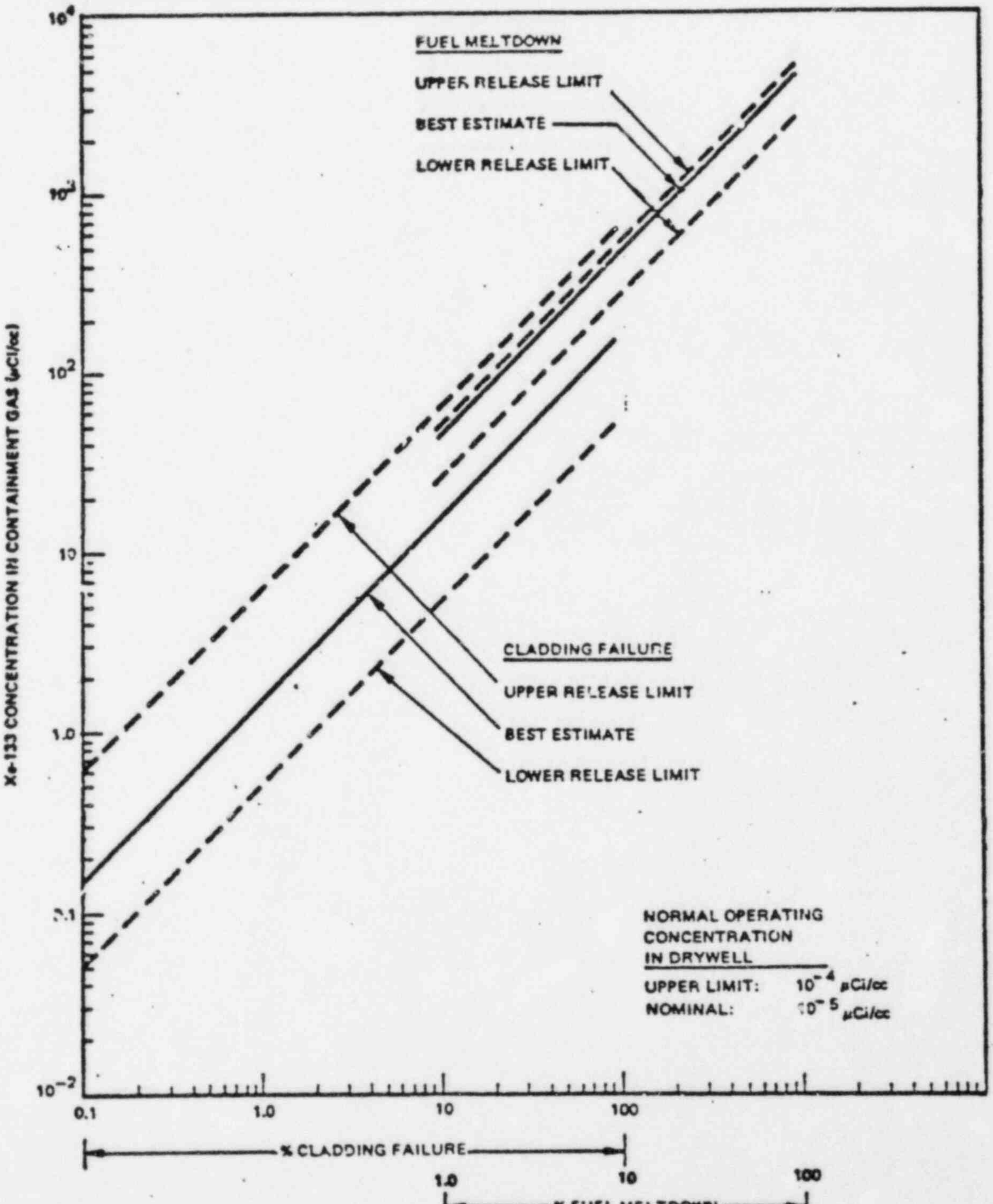
Relationship Between I-131 Concentration in the Primary coolant (Reactor Water + Pool Water) and the Extent of Core Damage in Reference Plant

EXHIBIT 3.6.3-2 (Cont'd)



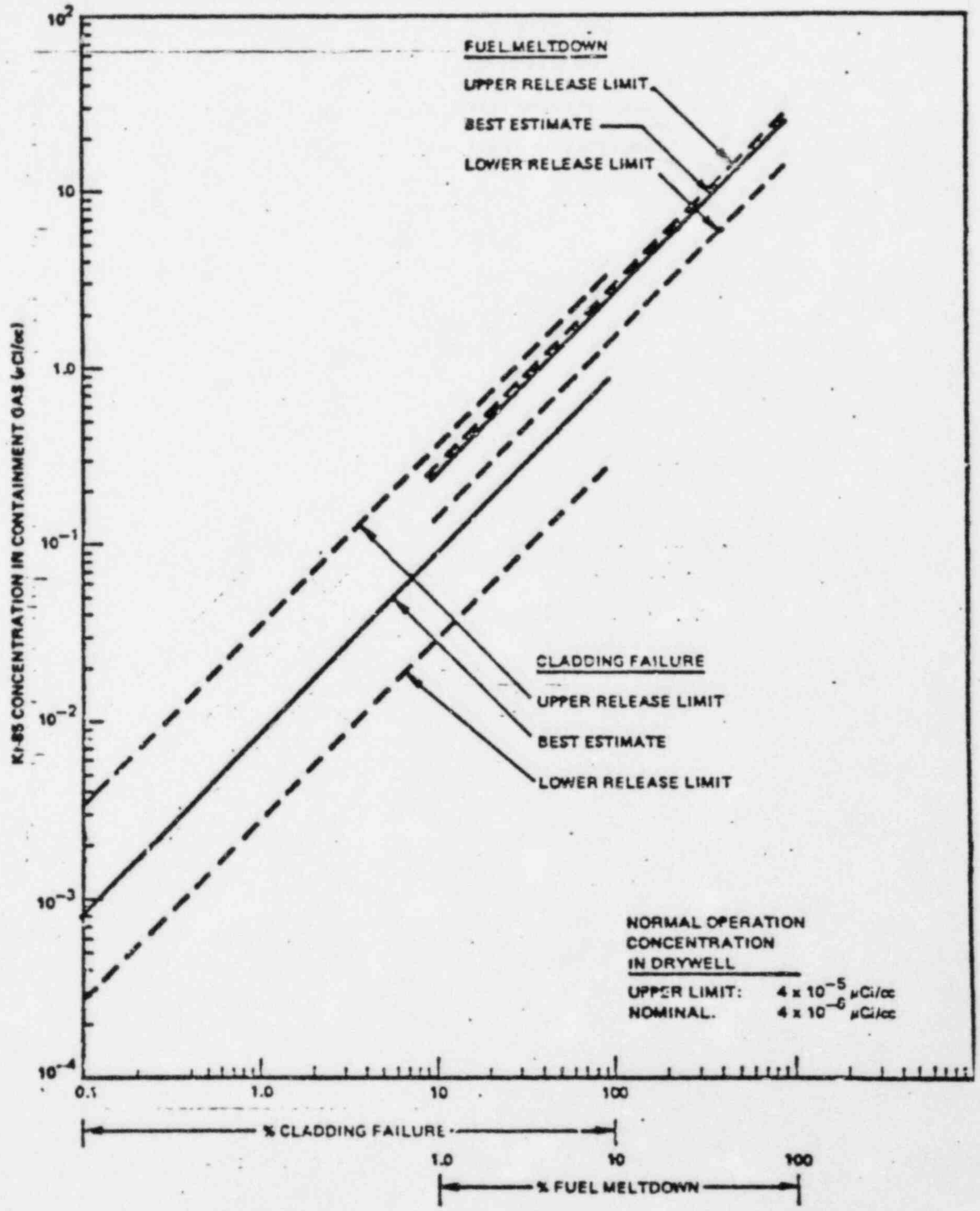
Relationship Between Cs-137 Concentration in the Primary Coolant (Reactor Water + Pool Water) and the Extent of Core Damage in Reference Plant.

EXHIBIT 3.6.3-2 (Cont'd)



Relationship Between Xe-133 Concentration in the Containment Gas (Drywell Torus Gas) and the Extent of Core Damage in Reference Plant.

EXHIBIT 3.6.3-2 (Cont'd)



Relationship Between Kr-85 Concentration in the Containment Gas (Drywell Torus Gas) and the Extent of Core Damage in Reference Plant.

APPENDIX A

Plant Parameter Correction Factors

Fission products measured together for reactor water and suppression pool water or drywell gas and torus gas.

$$F_w = \frac{\text{BSEP total coolant mass } (2.69 \times 10^9 \text{ g})}{\text{reference plant coolant mass } (3.92 \times 10^9 \text{ g})}$$

$$= 0.68622$$

$$F_g = \frac{\text{BSEP total containment gas volume } (8.11 \times 10^9 \text{ cc})}{\text{reference plant containment gas volume } (4 \times 10^{10} \text{ cc})}$$

$$= 0.20275$$

Fission products measured separately for reactor water and suppression pool water or drywell gas and torus gas.

$$C_{wi} = \frac{\text{(conc. in Rx wtr) (Rx water mass) + (conc. in pool) (pool wtr mass)}}{\text{reactor water mass + pool water}}$$

$$= \frac{\text{(conc. in Rx water)}(2.14 \times 10^8 \text{ g}) + \text{(conc. in pool)}(2.48 \times 10^9 \text{ g})}{2.69 \times 10^9 \text{ g}}$$

$$C_{gi} = \frac{\text{(conc. in drywell) (drywell gas vol.) + (conc. in torus) (torus gas vol.)}}{\text{drywell gas volume + torus gas volume}}$$

$$= \frac{\text{(conc. in drywell)}(4.65 \times 10^9 \text{ cc}) + \text{(conc. in torus)}(3.46 \times 10^9 \text{ cc})}{8.11 \times 10^9 \text{ cc}}$$

APPENDIX B

Inventory Correction Factor

$$F_{Ii} = \frac{\text{inventory in reference plant}}{\text{inventory in operation plant}}$$

$$= \frac{3651 \;\; 1-e^{-1095\lambda_i}}{\sum_j \left[P_j \;\; 1-e^{-\lambda_i T_j} \qquad e^{-\lambda_i T_j^{o}} \right]}$$

where:

P_j = average steady reactor power operated in period j (MWt).

T_j = duration of operating period j (day).

T_j^{o} = time between the end of operating period j and the time of the last reactor shutdown (day).

3651 = reference plant MWt.

If the unit operating history is not readily available, use the following F_I values (based upon Brunswick plant operations under the same operational constraints):

Nuclide	Conservative F_I	λ (day^{-1})
I-131	1.34	0.0862
Cs-137	1.39	6.29×10^{-5}
Xe-133	1.46	0.1320
Kr-85	1.51	1.77×10^{-4}

APPENDIX C

Comparison With Reference Plant Data

The extent of core damage can be estimated from the measure fission product concentrations in either the gas or water samples, as described for the reference plant. However, the measured concentration must be corrected for the differences in operation power level, time of operation, primary coolant mass, and containment gas volume.

$$C_{wi}^{Ref} = C_{wi} e^{\lambda_i t} \times F_{Ii} \times F_w$$

OR

$$C_{gi}^{Ref} = C_{gi} e^{\lambda_i t} \times F_{Ii} \times F_g$$

C_{wi}^{Ref} = Concentration of isotope i in the reference plant coolant (μCi/g).

C_{gi}^{Ref} = Concentration of isotope i in the reference plant containment gas (μCi/cc).

C_{wi} = Measured concentration of isotope i in BSEP's coolant (μCi/g). See Appendix A.

C_{gi} = Measured concentration of isotope i in BSEP's containment gas (μCi/cc). See Appendix A.

$e^{\lambda_i t}$ = Decay correction to the time of reactor shutdown.

λ_i = Decay constant of isotope i (day^{-1}).

t = Time between the reactor shutdown and the sample time (days).

F_{Ii} = Inventory correction factor for isotope i. See Appendix B.

F_g = Containment gas volume correction factor. See Appendix A.

F_w = Primary coolant mass correction factor. See Appendix A.

EXHIBIT 3.6.3-3

	Reference Plant	BSEP
Reactor Thermal Power	3651 MWt	2436 MWt
Number of Fuel Bundles	748 bundles	560 bundles
Total Primary Coolant Mass (reactor water plus suppression pool water)	3.92×10^{9} g	2.69×10^{9} g
Total Drywell and Torus Gas Space Volume	4.0×10^{10} cc	8.11×10^{9} cc
Reactor Water	2.46×10^{8} g	2.14×10^{8} g
Suppression Pool	3.67×10^{9} g	2.48×10^{9} g
Drywell Gas Volume	7.77×10^{18} cc	4.65×10^{9} cc
Torus Gas Volume	3.25×10^{10} cc	3.46×10^{9} cc

EXHIBIT 3.6.3-4

Core Inventory of Major Fission Products in a Reference Plant Operated at 3651 MWt for Three Years

Chemical Group	Isotope	Half-Life*	Inventory 10^6Ci	Major Gamma Ray Energy-Intensity - keV(γ/d)
Noble Gases	Kr-85m	4.48 h	24.6	151 (0.753)
	Kr-85	10.72 y	1.1	514 (0.0044)
	Kr-87	76.00 m	47.1	403 (0.495)
	Kr-88	2.84 h	66.8	196 (0.26), 1530 (0.109)
	Xe-133	5.25 d	202.0	81 (0.365)
	Xe-135	9.09 h	26.1	250 (0.899)
Halogens	I-131	8.04 d	96.0	364 (0.812)
	I-132	2.29 h	140.0	668 (0.99), 773 (0.762)
	I-133	20.80 h	201.0	530 (0.86)
	I-134	52.60 m	221.0	847 (0.954), 884 (0.653)
	I-135	6.59 h	189.0	1132 (0.225), 1260 (0.286)
Alkali Metals	Cs-134	2.06 y	19.6	605 (0.98), 796 (0.85)
	Cs-137	30.17 y	12.1	662 (0.85)
	Cs-138	32.20 m	2990.0	463(0.307), 1436 (0.76)
Tellurium Group	Te-132	78.00 h	138.0	228 (0.88)
Noble Metals	Mo-99	66.02 h	183.0	740 (0.128)
	Ru-103	39.40 d	155.0	497 (0.89)
Alkaline Earths	Sr-91	9.52 h	115.0	750 (0.23), 1024 (0.325)
	Sr-92	2.71 h	123.0	1384 (0.9)
	Ba-140	12.80 d	173.0	537 (0.254)
Rare Earth	Y-92	58.60 d	118.0	934 (0.139)
	La-140	40.20 h	184.0	487 (0.455), 1597 (0.955)
	Ce-141	32.50 d	161.0	145 (0.48)
	Ce-144	284.40 d	129.0	134 (0.108)
Refractories	Zr-95	46.00 d	161.0	724 (0.437), 757 (0.553)
	Zr-97	16.80 h	166.0	743 (0.928)

* h = hour
d = day
m = month
y = year

APPENDIX D

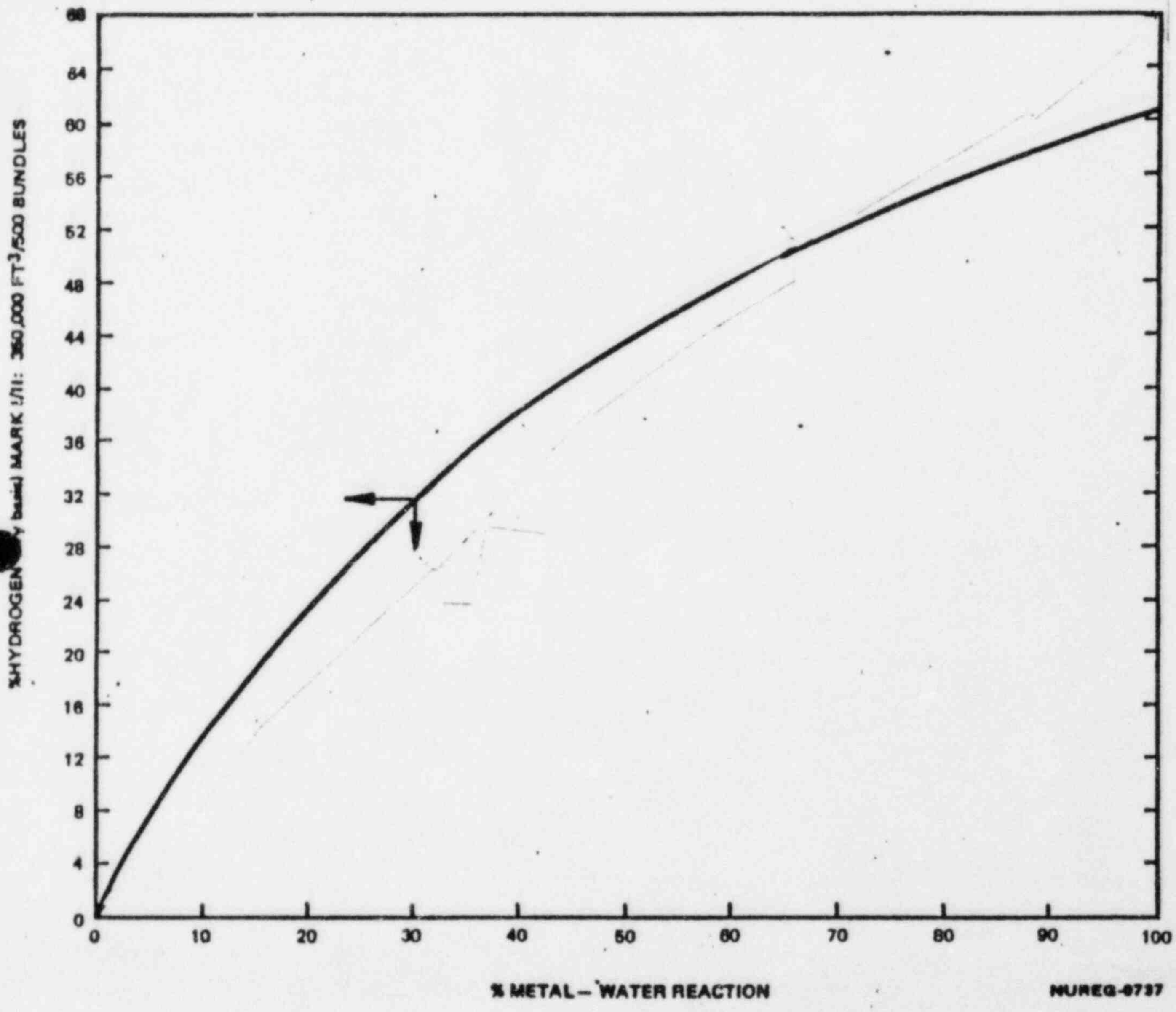
Integration of Containment Atmosphere Hydrogen Measurement Into Core Damage Estimate

The extent of fuel clad damage as evidenced by the extent of metal-water reaction can be estimated by determination of the hydrogen concentration in the containment. That concentration is measurable by either the containment hydrogen monitor or by the Postaccident Sampling System.

A correlation has been developed which relates containment hydrogen concentration to the percent metal-water reaction for Marks I and II type containments. That correlation is shown in Exhibit 3.6.3-5. Note A to that exhibit indicates the major assumptions used in developing the correlation. Note B indicates the method by which Brunswick plant can use the correlation to determine the extent of clad damage.

APPENDIX D (Cont'd)

EXHIBIT 3.6.3-5



Hydrogen Concentration for Marks I and II Containments as a
Function of Metal-Water Reaction

APPENDIX D (Cont'd)

Note A to Exhibit 3.6.3-5
Analytical Assumptions
(For Marks I and II Containments)

1. Containment Volume = 350,000 ft^3

2. Number of Bundles = 500

3. Fuel Type = 8 x 8 R

4. All hydrogen from metal-water reaction released to containment.

5. Perfect mixing in containment.

6. No depletion of hydrogen (e.g., containment leakage).

7. Ideal gas behavior in containment.

APPENDIX D (Cont'd)

Note B to Exhibit 3.6.3-5

Determination of Clad Damage From Hydrogen Monitor Reading

Step 1. Obtain containment hydrogen monitor reading in percent.

Step 2. Using the curve in Exhibit 3.6.3-5, determine the metal-water reaction for the reference plant, MWR_{ref}.

Step 3. The metal-water reaction from the actual in-plant conditions (MWR) in determined from the following equation:

$$\% \ MWR = (MWR_{ref}) \times \frac{500}{N} \times \frac{V}{350{,}000}$$

where:

N = Number of Bundles = 560
V = Total Containment Free Volume, ft^3 = 2.86×10^5

APPENDIX E

Integration of Containment Atmosphere Radiation Measurement Into Core Damage Estimate

An indication of the extent of core damage is the containment radiation level which is a measure of the inventory of fission products released to the containment. This appendix contains a correlation of the containment radiation monitor dose rate to the percent of fuel inventory airborne in the containment. The purpose of this appendix is to present that correlation and provide a method to use that correlation to determine the degree of core damage.

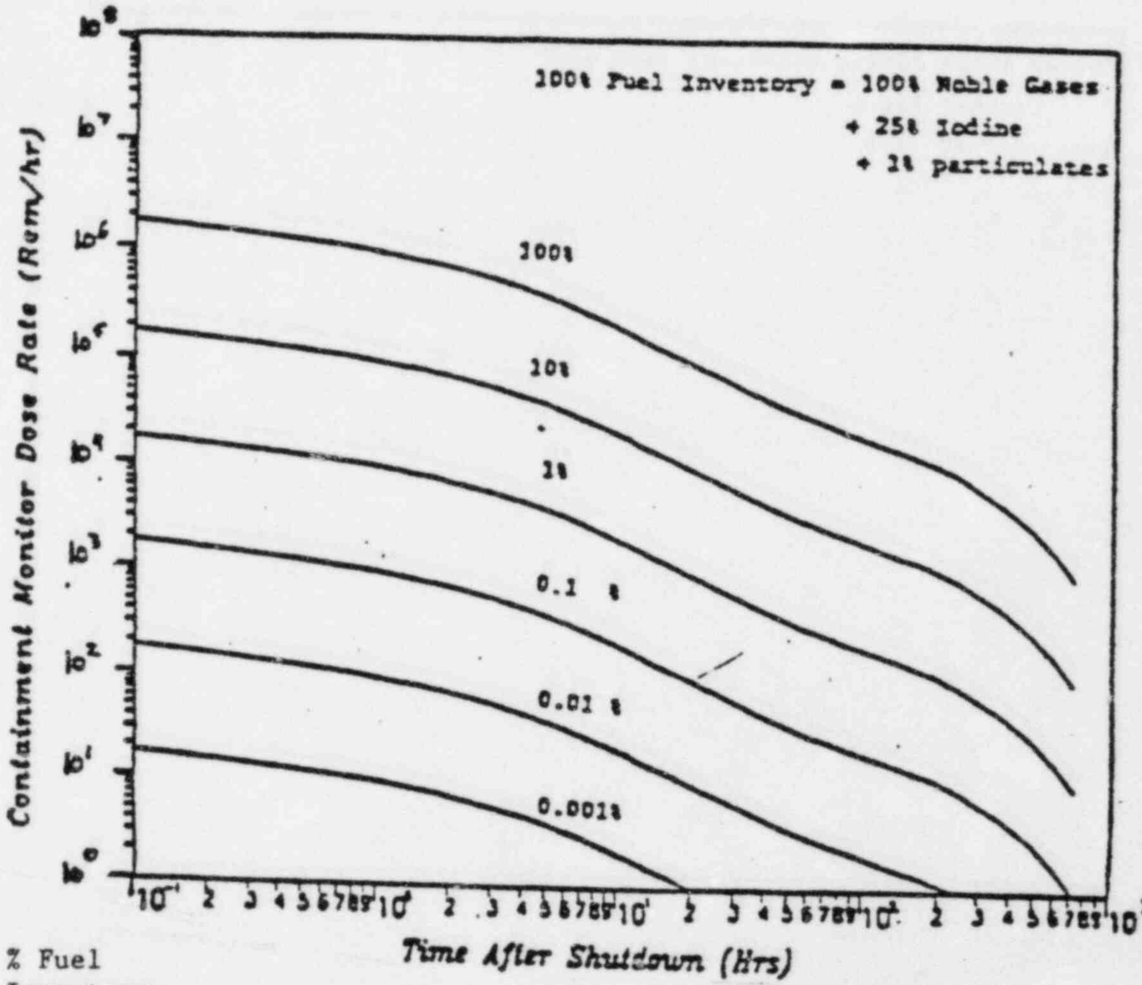
Exhibit 3.5.3-6 provides the results of a correlation performed for the Monticello plant. The key parameters which impact the containment dose rate are reactor power, containment volume, and monitor location within the containment.

The method whereby individual plants can apply this correlation is provided in Note A to Exhibit 3.6.3-6.

APPENDIX E (Cont'd)

APPENDIX 3.6.3-6

Percent of Fuel Inventory Airborne in the Containment



% Fuel Inventory Released	Approximate Source and Damage Estimate
100.00	100% TID-14844, 100% fuel damage, potential core melt.
50.00	50% TID noble gases, TMI source.
10.00	10% TID, 100% NRC gap activity, total clad failure, partial core uncovered.
3.00	3% TID, 100% WASH-1400 gap activity, major clad failure.
1.00	1% TID, 10% NRC gap, maximum 10% clad failure.
0.10	0.1% TID, 1% NRC gap, 1% clad failure, local beating of 5-10 fuel assemblies.
0.01	0.01% TID, 0.1% NRC gap, clad failure of 3/4 fuel element (36 rods).
10^{-3}	0.01% NRC gap clad failure of a few rods.
10^{-4}	100% coolant release with spiking.
5×10^{-6}	100% coolant inventory release.
10^{-6}	Upper range of normal airborne noble gas activity in containment.

APPENDIX E (Cont'd)

NOTE A to Exhibit 3.6.3-6

Determination of Clad Damage From Containment
Radiation Monitor Reading

The procedure for determination of fraction of fuel inventory released to the containment is as follows:

Step 1: Obtain containment radiation monitor reading, [R] in rem/hr.

Step 2: Determine elapsed time from plant shutdown to the containment radiation monitor reading [t] in hours.

Step 3: Using Exhibit 3.6.3-6, determine the fuel inventory release for the reference plant $[I]_{ref}$ in percent.

Step 4: Determine the inventory release to the containment [I] using the following formula:

$$[I] = [I]_{ref} \left(\frac{1670}{P}\right)\left(\frac{V}{237,450}\right)(6/D)$$

where:

P = reactor power level MW_{th} (BSEP = 2436 MW_{th}).

V = total containment free volume, ft^3 (BSEP = 286, 370 ft^3).

D = distance of detector from reactor biological shield wall, ft.

EXHIBIT 3.6.3-7

Computer Inputs for The PASS Program

Concentration of I-131 in Reactor Water (µCi/ml)	________
Concentration of I-131 in Suppression Pool (µCi/ml)	________
Concentration of Cs-137 in Reactor Water (µCi/ml)	________
Concentration of Cs-137 in Suppression Pool (µCi/ml)	________
Concentration of Xe-133 in Drywell (µCi/cc)	________
Concentration of Xe-133 in Torus (µCi/cc)	________
Concentration of Kr-85 in Drywell (µCi/cc)	________
Concentration of Kr-85 in Torus (µCi/cc)	________
Time between Reactor Shutdown and Sample Time (days)	________

If time and availability permits, attach information necessary for the calculation of Inventory Correction Factors (see Appendix B); otherwise, conservative default correction factors will be used.

Plant Sampling and Analysis Team Leader: Give completed exhibit to Dose Projection Coordinator.

Dose Projection Coordinator: Enter data into PASS computer program and provide results to Plant Sampling and Analysis Team Leader.

EXHIBIT 3.6.3-8

VERIFICATION OF PASS

(A Computer program for estimating core damage
based on Postaccident Sampling System results)

This exhibit is intended to provide a means to ensure that PASS, a core damage estimate program designed for the IBM Personal Computer, is working properly. This is demonstrated by duplicating expected results of known computer inputs. These results can be validated by comparison to manual calculations for the same input.

Two different test cases are presented so that a number of alternate paths within the program can be tested. The test cases with their expected results follow.

TEST CASE 1

Computer Prompt	Expected Input
Enter The Concentration of the Fission Products	
Concentration of I-131 in Reactor Water (μCi/ml)	1.72E + 3
Concentration of I-131 in Suppression Pool (μCi/ml)	1.49E + 2
Concentration of Cs-137 in Reactor Water (μCi/ml)	6.55E + 2
Concentration of Cs-137 in Suppression Pool (μCi/ml)	5.70E + 1
Concentration of Xe-133 in Drywell (μCi/cc)	1.82E + 2
Concentration of Xe-133 in Torus (μCi/cc)	2.41E + 2
Concentration of Kr-85 in Drywell (μCi/cc)	1.43E + 0
Concentration of Kr-85 in Torus (μCi/cc)	1.90E + 0

For the inventory correction factor do you want to use the conservative default values which are bases upon BSEP's operations under the same operational constraints (YES or NO)? YES

Enter time between the reactor shutdown and the Sample Time (Days) 2

The results should resemble the printout on the following page. If they do not, carefully check your inputs and try the test again. If the results still are not similar, try a backup copy of the program. If that fails, then seek programming help.

EXHIBIT 3.6.3-8 (Cont'd)
ESTIMATE THE EXTENT OF CORE DAMAGE UNDER ACCIDENT CONDITIONS

DATE: 03-28-1984
TIME: 13:21:27

The concentration of the fission products are:

I-131 in Reactor Water	1.72E + 3 μCi/ml
I-131 in Suppression Pool	1.49E + 2 μCi/ml
Cs-137 in Reactor Water	6.55E + 2 μCi/ml
Cs-137 in Suppression Pool	5.70E + 1 μCi/ml
Xe-133 in Drywell Air	1.82E + 2 μCi/cc
Xe-133 in Torus Air	2.41E + 2 μCi/cc
Kr-85 in Drywell Air	1.43E + 0 μCi/cc
Kr-85 in Torus Air	1.90E + 0 μCi/cc

Time between the reactor shutdown and the sample time is: 2 days

The Concervative Default values of the Inventory Correction Factors were used.

Estimate of fuel/cladding damage
Primary Coolant Analysis

Nuclide	CwREF (μCi/ml)	% Cladding Failure	% Fuel Meltdown
I-131	3.00E + 02	69.00	1.35
Cs-137	1.00E + 02	64.54	4.27

Containment Gas Analysis

Nuclide	CwREF (μCi/ml)	% Cladding Failure	% Fuel Meltdown
Xe-133	7.99E + 01	53.26	1.84
Kr-85	5.00E - 01	56.35	1.92

EXHIBIT 3.6.3-8 (Cont'd)

TEST CASE 2

Computer Prompt	Expected Input
Enter The Concentration of the Fission Products	
Concentration of I-131 in Reactor Water (µCi/ml)	1.35E + 3
Concentration of I-131 in Suppression Pool (µCi/ml)	1.18E + 2
Concentration of Cs-137 in Reactor Water (µCi/ml)	1.17E + 2
Concentration of Cs-137 in Suppression Pool (µCi/ml)	1.02E + 1
Concentration of Xe-133 in Drywell (µCi/cc)	1.84E + 2
Concentration of Xe-133 in Torus (µCi/cc)	2.45E + 2
Concentration of Kr-85 in Drywell (µCi/cc)	2.91E - 1
Concentration of Kr-85 in Torus (µCi/cc)	3.86E - 1
For the inventory correction factor do you want to use the conservative default values which are bases upon BSEP's operations under the same operational constraints (YES or NO)?	NO
Enter time between the reactor shutdown and the Sample Time (Days)?	2
Enter number of Operating Periods from the unit operating history?	3
For period number (1) enter:	
Average steady reactor power operated in this period (MWT)?	1000
Duration of this operating period (days)?	60
Time between the end of this operating period and the time of the most recent reactor shutdown (days)?	254
For period number (2) enter:	
Average steady reactor power operated in this period (MWT)?	2000
Duration of this operating period (days)?	200
Time between the end of this operating period and the time of the most recent reactor shutdown (days)?	44
For period number (3) enter:	
Average steady reactor power operated in this period (MWT)?	3000
Duration of this operating period (days)?	14
Time between the end of this operating period and the time of the most recent reactor shutdown (days)?	0

The results should resemble the printout on the following page. If they do not, carefully check your inputs and try the test again. If the results still are not similar, try a backup copy of the program. If that fails, then seek programming help.

EXHIBIT 3.6.3-8 (Cont'd)
ESTIMATE THE EXTENT OF CORE DAMAGE UNDER ACCIDENT CONDITIONS
DATE: 03-28-1984
TIME: 13:27:17

The concentration of the fission products are:

I-131 in Reactor Water	1.35E + 3 µCi/ml
I-131 in Suppression Pool	1.18E + 2 µCi/ml
Cs-137 in Reactor Water	1.17E + 2 µCi/ml
Cs-137 in Suppression Pool	1.02E + 1 µCi/ml
Xe-133 in Drywell Air	1.84E + 2 µCi/cc
Xe-133 in Torus Air	2.45E + 2 µCi/cc
Kr-85 in Drywell Air	2.91E - 1 µCi/cc
Kr-85 in Torus Air	3.86E - 1 µCi/cc

Time between the reactor shutdown and the sample time is: 2 days

The Inventory Correction Factors were calculated from the following:

Period No.	Operation Time (days)	Time Between Period & Last Shutdown (days)	Average Power (MWt)
1	60	254	1000
2	200	44	2000
3	14	0	3000

Estimate of Fuel/Cladding Damage
Primary Coolant Analysis

Nuclide	CwREF (µCi/ml)	% Cladding Failure	% Fuel Meltdown
I-131	3.00E + 02	69.02	1.35
Cs-137	9.99E + 01	64.49	4.27

Containment Gas Analysis

Nuclide	CwREF (µCi/ml)	% Cladding Failure	% Fuel Meltdown
Xe-133	8.00E + 01	53.30	1.84
Kr-85	5.00E - 01	56.40	1.92

WORKSHEET A1

CALCULATION OF ISOTOPIC CONCENTRATIONS IN PRIMARY WATER AND SUPPRESSION POOL WATER (Cw_i) AND DRYWELL GAS AND TORUS GAS (Cg_i)

References

Section 3.3.1.2
Section 3.3.1.5
Appendix A
Exhibit 3.6.3-3

Cw_i (µCi/ml) = (Concentration Rx H_2O)$_i$ (0.08) +
(Cs^{137}, I^{131}) (Concentration Suppression Pool H_2O)$_i$ (0.92)

= [____________________ + ____________________] (µCi/ml)

= __________ µCi/ml$_{Cs^{137}}$

and = __________ µCi/ml$_{I^{131}}$

Cg_i (µCi/ml) = (Concentration Drywell)$_i$ (0.57) + (Concentration Torus)$_i$ (0.43)
(Xe^{133}, Kr^{85})

= [____________________ + ____________________] (µCi/cc)

= __________ µCi/cc$_{Xe^{133}}$

and = __________ µCi/cc$_{Kr^{85}}$

WORKSHEET A2

CALCULATION OF INVENTORY CORRECTION FACTOR (FI_i)

References

Section 3.3.1.4
Appendix B
Exhibit 3.6.3-4

P_j = __________ $MW_{thermal}$

$T_{j°}$ = __________ Days

T_j = __________ Days

λ_i = __________ $Days^{-1}$

$$FI_i = \frac{3651\ (1 - e^{-1095\lambda_i})}{\Sigma_j \left[P_j\ (1 - e^{-\lambda_i T_j})(e^{-\lambda_i T_{j°}})\right]}$$

= ______________________________ (Cs^{137})

______________________________ (I^{131})

______________________________ (Xe^{133})

______________________________ (Kr^{85})

WORKSHEET A3

CALCULATION OF NORMALIZED ISOTOPIC CONCENTRATIONS IN PRIMARY WATER AND SUPPRESSION POOL WATER (Cw_i^{Ref}) AND DRYWELL GAS AND TORUS GAS (Cg_i^{Ref})

References

Section 3.3.1.6
Appendix C
Worksheet A1
Worksheet A2

NOTE: For BSEP,
$Fw = 0.68622$
$Fg = 0.20275$

$$Cw_i^{Ref} = Cw_i\, e^{\lambda i t} \times FI_i \times Fw$$
(Cs^{137}, I^{131})

= ______________________________ $\mu Ci/ml_{Cs^{137}}$

______________________________ $\mu Ci/ml_{I^{131}}$

$$Cg_i^{Ref} = Cg_i\, e^{\lambda i t} \times FI_i \times Fg$$
(Xe^{133}, Kr^{85})

= ______________________________ $\mu Ci/cc\ Xe^{133}$

______________________________ $\mu Ci/cc\ Kr^{85}$

WORKSHEET A4

ESTIMATE OF FUEL/CLADDING DAMAGE

References

Section 3.3.1.7
Exhibit 3.6.3-2
Worksheet A3

Primary Coolant Analysis

Isotope	Cw_i^{Ref} (μCi/ml)	% Cladding Failure	% Fuel Meltdown
I^{131}			
Cs^{137}			

Containment Gas Analysis

Isotope	Cg_i^{Ref} (μCi/ml)	% Cladding Failure	% Fuel Meltdown
Xe^{133}			
Kr^{85}			

WORKSHEET B1

DETERMINATION OF CLAD DAMAGE FROM HYDROGEN MONITOR READING

References

Section 3.4.1
Appendix D
Exhibit 3.6.3-5

Containment Hydrogen Monitor Reading: ________________%

MWR ref : ________________%

Calculate % MWR:

% MWR = (MWR ref)(0.73)
= ______________

WORKSHEET B2

DETERMINATION OF FUEL INVENTORY RELEASE BASED ON CONTAINMENT RADIATION MONITOR READING

References

Section 3.4.2
Appendix E
Exhibit 3.6.3-6

NOTE: D = Distance of Radiation Monitor from Biological Shield Wall, ft.

Containment Radiation Monitor Reading: ____________________ Rem/hr

Time from Shutdown to Monitor Reading: ____________________ hrs

[I] ref (Reference Fuel Inventory Release, %, from Exhibit 3.6.3-6) : ____________________ %

I (Actual Fuel Inventory Release) = [I] ref $\frac{4.96}{D}$

= ____________ %

CAROLINA POWER & LIGHT COMPANY
BRUNSWICK STEAM ELECTRIC PLANT

UNIT 0

COLLECTION AND ANALYSIS OF VERY HIGH LEVEL RADIOACTIVE SAMPLES

PLANT EMERGENCY PROCEDURE: PEP-03.6.5

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General Manager

LIST OF EFFECTIVE PAGES

PEP-03.6.5

Page(s)	Revision
1-3	0

1.0 Responsible Individual and Objectives

The Plant Sampling and Analysis Team is responsible to the Radiological Control Director for obtaining and analyzing required very high level radioactive samples by utilizing the proper sample equipment, protective clothing, and collection methods.

2.0 Scope and Applicability

This procedure shall be implemented as directed by the Site Emergency Coordinator or Radiological Control Director. Very high level radioactive samples are so designated if contact dose rate levels exceed 2.0 R/hr.

3.0 Actions and Limitations

CAUTION: If gross failure of cladding occurs, significant quantities of noble gases or other volatiles, as well as other fission products, may be released. Levels to the order of 10,000 μCi/ml may be present in the sample media, thus the usual laboratory analysis procedures may be inadequate for processing such samples.

3.1 The Plant Sampling and Analysis Team Leader shall, as necessary, specify to the Plant Sampling and Analysis Team:

3.1.1 Procedures to be carried out, such as:

3.1.1.1 E&RC-1500, Analysis of PASS Samples in the Laboratory.

3.1.1.2 E&RC-1501, Alternate Emergency Sampling of Drywell and Torus Gas Using CAC-1259 and CAC-1263.

3.1.1.3 E&RC-1502, Emergency Sampling of Reactor and Turbine Building Ventilation Monitors.

3.1.1.4 E&RC-1503, Emergency Sampling of Stack Monitor.

3.1.1.5 E&RC-1504, Postaccident Analysis by Ion Chromatography.

3.1.1.6 E&RC-1505, Operation Procedure for Postaccident Sampling Stations.

3.1.1.7 E&RC-1520, Analysis of Gaseous Samples After a Fuel Element Accident.

3.1.2 Protective gear and communication equipment required.

3.1.3 Alternate entry and egress routes.

3.1.4 Alternate analysis locations.

3.1.5 Types of samples to be collected.

3.1.6 Special radiation safety precautions for the handling and disposal of samples.

3.1.7 Exceptions to routine plant procedures.

3.1.8 Format for results to be presented in.

3.1.9 Have backup teams ready as necessary.

3.2 The Plant Sampling and Analysis Team shall:

3.2.1 Carry out very high sample collections per the plant Sampling and Analysis Team Leader's instructions.

3.2.2 Minimize radiation exposures by effective use of barriers (shielding), reduced stay times, and increased distances from sources.

3.2.3 Assure that each sample container is labeled with:

3.2.3.1 Name and type of sample;

3.2.3.2 Sample time;

3.2.3.3 Sample number, if applicable;

3.2.3.4 Sample location, and;

3.2.3.5 Contact dose rate of sample.

3.2.4 Document on appropriate counting room forms:

3.2.4.1 All procedures utilized to accomplish a particular sample and analysis;

3.2.4.2 Exceptions and inclusions to be used and/or partially utilized procedures, and;

3.2.4.3 All analyses results.

3.2.5 If local analysis facilities become contaminated or otherwise unusable:

3.2.5.1 Contact the Plant Sampling and Analysis Team Leader and request notification of off-site laboratories at Robinson, at Shearon Harris Energy and Environmental Center Laboratories or Babcock and Wilcox (see Appendix A) and/or a contracted radioactive material shipper to assist in analysis and shipping.

<u>NOTE</u>: Inform the Plant Sampling and Analysis Team Leader of the required schedule for results.

3.2.5.2 Containerize the sample in accordance with standard plant procedures for shipping radioactive samples.

3.2.5.3 Ship the sample, the required sample information, shipping information, and required results to the selected off-site laboratory.